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INTR

CONSOLIDATED EDISON COMPANY

OF NEW YORK, INC.

INDIAN POINT UNIT NO. 2

FURTHER ANSWERS TO QUESTIONS

FROM THE CITIZEN'S COMMITTEE

OF PROTECTION TO THE ENVIRONMENT

FEBRUARY 5, 1971

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Question No. C-7

Question: An explosive rupture of the steam turbine high pressure or cross-over piping or casings can occur, as it has in other power stations. It can cause or be associated with a generator failure and consequent hydrogen fire. Can such an explosion and fire in any way result in a rupture of the containment vessel? Or could a sequence be a logical consequence of a non-explosive primary system failure inside the containment vessel?

Answer: The explosion or fire such as described cannot result in a rupture of the containment vessel, nor can such a sequence be a logical consequence of a non-explosive primary system failure (See FSAR Section 14.3 and Appendix 14A).

Question No. D-2

Question: Why does the AEC require a 50-mile radius population analysis?

Answer: (AEC response).

Question No. D-3

Question: What safety features do two or more of the Indian Point reactors share?

Answer: As stated in the Summary of Application, Unit No. 2 does not share safety-related systems with either of the other two units on the site. Units 1 and 2 will have a common control room, but the controls for each unit are physically separate. The three units also have a common discharge canal, and there are certain other ties between them such as backup electrical power supplies, city water, and sanitary facilities. Unit No. 2 is therefore virtually independent of the other two units.

Question No. D-11

Question: For "allowable emissions", describe which have been arrived at by calculations, which by assumptions, which by actual testing of comparable equipment?

Answer: "Allowable emissions" for the Indian Point site are set according to the requirements set forth in 10CFR20 for normal operating releases.

These requirements are satisfied if the following criteria are met:

Gaseous Releases

1) An individual standing at any point on or beyond the the restricted area will not receive an annual whole body dose greater than 0.5 rem.

2) The annual average ground level concentrations of the released isotopes will not exceed the restrictions of 10CFR20.106 at any point on or beyond the restricted area.

Liquid Releases

3) The concentration of released isotopes in the outfall of the facility (three plants) prior to entrance into the river will not exceed the limits set forth in 10CFR20.

A complete description of how the provisions of 10CFR20 (and the aforementioned criteria) are met for the Indian Point site (three plants) is presented as part of the answer to AEC Question 11.1 (Supplement 15, 11/70).

The technical specifications for Indian Point Unit No. 2 place further restrictions on "allowable emissions" consisting primarily of a requirement that instantaneous release rates are not allowed to exceed the average annual release rates allowed by 10CFR20.

The method of determining releases is to measure the quantity of radio activity which is to be released and then calculating the release rate at which it can be released to be in accordance with 10CFR20 and the Technical Specifications.

Question No. D-15

Question: Since the AEC has ruled that for purposes of "allowable emissions", multiple reactors must be treated as a unit and Techncial Specifications reflect the effects of combined operations, why for siting guidelines should not the combined power rating of a complex of reactors be taken into account?

Answer: (AEC response).

Question No. D-16

Question: How many "engineered" safeguards have been devised to attempt to achieve the safety factor of isolation?

Answer: The five systems listed in the response to Set A Question 9, in conjunction with the containment, achieve the safety factor of isolation.

Question No. D-19

Question: Define the term "engineered".

Answer: (AEC response).

Question No. D-20

Question: Explain the system by which "credits" are given for engineered safeguards?

Answer: (AEC response).

Question No. D-21

Question: Furnish authorized reports on the range of credits accorded for each safeguard.

Answer: (AEC response).

Question No. D-22

Question: How can these credits be evaluated in the light of 10CFR100?

Answer: (AEC response).

Question No. D-24

Question: What percent of each safeguard can qualify as "engineered"

Answer: (AEC response).

Question No. D-26

Question: Prepare a table for all Pressurized-Water Reactors with the credits given for various engineered safeguards and justifications for any differences.

Answer: (AEC response).

Question No. D-27

Question: Assuming an inoperative emergency core cooling system, describe the sequence of events.

Answer: During operation, an inoperative ECCS or any inoperative vital components of that ECCS which cannot be restored to operation would require the plant to be shut down in accordance with the proposed Technical Specifications.

The core cooling systems are designed to be redundant so that the requirements of core cooling during or after an accident may be met even if one of the redundant components is inoperative.

Question No. D-36

Question: Has the AEC furnished Con Edison a calculation of the genetic damage to the national germ pool from the number of "burned-out" employees at all AEC licensed installations?

Answer: (AEC response).

Question No. D-39

Question: What provisions are there for in-service inspections?

Answer: The complete In-Service Inspection program is detailed in Section 4.2 of the proposed Indian Point Unit No. 2 Technical Specifications.

Specific provisions for in-service inspection are detailed in Section 4.5, page 4.5-6 through page 4.5-10. Generally, these provisions consist of designed access, removable insulation, and pre-service base reference inspections.

Question No. D-41

Question: Furnish a map of known faults up to three inches and list the tests results of ultrasonic and other techniques.

Answer: There are no faults in the Indian Point Unit No. 2 reactor pressure vessel since no indications exceeded code requirements. The test results of ultrasonic and other techniques demonstrate compliance with the code. Underlying documents which are voluminous, bulky, and in part non-reproducable with respect to such results are in the safe keeping of the vessel manufacturer Combustion Engineering at Chattanooga, Tennessee.

Question No. D-43

Question: Furnish stress analysis of pressure vessel.

Answer: A copy of the stress analysis of the pressure vessel is available for inspection. However, a summary description of the stress analysis is set forth in response to Question 4.8.1 in Volume 5 of the FSAR.

Question No. D-44

Question: Furnish inspections and radiographs reports on all welds.

Answer: These documents, which are voluminous, bulky and in part non-reproducable, are in the safe keeping of the vessel manufacturer Combustion Engineering at Chattanooga, Tennessee.

Question No. D-45

Question: Is there a map of all known faults in the "as constructed pressure vessel?

Answer: See answer to Set D Question 41.

Question No. D-46

Question: How will fault growth be monitored?

Answer: There are no faults in the "as constructed" pressure vessel (See answer to Set D Question 41). Indications are identified with a base line ultrasonic map. In-service ultrasonic mapping will provide information on the behavior of the indications. See Technical Specification 4.2.

Question No. D-47

Question: Describe what precautions will be taken to be alerted to possibility of brittle fracture.

Answer: See discussion on Surveillance Requirements in Technical Specification 4.2.

Question No. D-48

Question: Describe the procedure for using samples in the reactor to see how much neutron bombardment causes losses in strength.

Answer: Neutron bombardment does not decrease the tensile strength properties of the vessel material. See answer to Set D Question 47 for surveillance sample withdrawal program.

Question No. D-49

Question: Describe the operation of the low-pressure coolant following a double-ended pipe rupture and the expulsion of all coolant water.

Answer: The details of the actuation and operation of the lowhead safety injection sub-system is described in the FSAR, Section 6.2.

Question No. D-52

Question: Has a functional test been conducted of the ECCS under design basis accident conditions?

Answer: See responses to Set A Question No. 13, Set F Question 34, and FSAR answer to Question 7.8.

Question No. D-53

Question: What scheduled tests of adequacy of components remain to be done - or are awaiting completion of LOFT facilities?

Answer: None are awaiting completion of LOF T facilities. Those tests remaining to be performed at Indian Point Unit No. 2 are given in the Testimony of J. T. Stiefel, already introduced into evidence (Transcript Page 384), Pages 5 and 6.

Question No. D-54

Question: When will LOFT be completed for testing and at what expenditure?

Answer: (AEC response).

Question No. D-55

Question: Justify the risk to public, depending on ECCS, which cannot be functionally tested?

Answer: (AEC response).

Question No. D-58

Question: What documents are available indicating inspections that Con Edison or its vendors have done on parts?

Answer: The FSAR lists those inspections that Con Ed or its vendor has performed on components of the Indian Point #2 plant. Specifically, locations of the inspections for each component are listed in the section of the FSAR describing the components (Inspection and Tests);(e.g., Table 4.5-1 for the Reactor Coolant System Components, Table 6.2-13 for the ECCS components). Appendix B of the FSAR indicates overall the maintenance of records of inspection and associated quality assurance.

Question No. D-60

Question: What official at Con Edison is familiar with all the steps that have been taken to see that quality control has been exercised over all components and systems?

Answer: No single person is familiar with all the steps that have been taken. Mr. J. J. Grob, Jr., Chief Mechanical Engineer, is the Con Edison representative who is familiar with the overall quality control steps that have been exercised over the plant components and systems.

Question No. D-66

Question: What would be the result, if the secondary coolant from the river were to stop abruptly?

Answer: The result of loss of "secondary coolant" could be either the loss of circulating water to the main condensers or loss of service water. Both contingencies are considered in the design of Indian Point Unit No. 2 and the analysis of what happens is found in the Indian Point Unit No. 2 FSAR, Section 14.1.9, Appendix 14A, Section 5.2 These analyses show that the loss of either circulating water or service water or both would result in plant shutdown and maintenance of the plant in a safe shutdown condition.

Question No. D-68

Question: At what temperature range will core spray system sputter?

Answer: Indian Point Unit No. 2 does not have a core spray system.

Question No. D-69

Question: What provisions have been made to prevent all the molten uranium from accumulating as a mass at the bottom of the reactor, trapping water, and resulting in a steam explosion?

Answer: As set forth in the answer to Set A Question 8, major meltdown, and hence the existence of molten uranium, is not a postulated accident for this plant. An Emergency Core Cooling System has been installed in the Indian Point Unit No. 2 plant which maintains core temperature below the point where the fuel melt can occur.

Question No. D-70

Question: Furnish Applicant's Quality Assurance Codes, where does it differ from AEC's.

Answer: The word "code" is inappropriate, since neither the Con Edison nor the AEC has a Quality Assurance Code.

Con Edison quality Assurance Program for Indian Point Unit No. 2 is contained in the FSAR, Appendix B. This program has been reviewed and approved by the AEC.

Question No. D-72

Question: Furnish a schedule of quality assurance documents that will be prepared.

Answer: Principal quality assurance documents that have been prepared for IP-2 are described in the FSAR Appendix B.

Question No. D-80

Question: What will be the route of the carriers conveying the high-level wastes to reprocessing?

Answer: See response to Question 56, Set A.

Question No. D-81

Question: How many shipments will be made and at what intervals?

Answer: On the average, there will be thirty-two (32) truck shipments each year occurring within an eight week period from Unit No. 2

Question No. D-83

Question: Has a probability study been run on the possibility of accidents to high level waste in transit.

Answer: Applicant is not aware of any probability studies made on accidents to high level waste (nuclear fuel) in transit.

Question No. D-84

Question: What will be the schedule when wastes from No. 1, 3 4, and 5 are operable?

Answer: The schedule for IP-2 will be the same as presented in response to Q. 81, set D.

On the average there will be 52 truck shipments made from IP-1 and IP-3 each year. The waste shipment schedule has not yet been studied for the proposed Nuclear Units 4 and 5 (Verplanck 1 and 2).

Question No. D-92

Question: Furnish a trade-off and safety study of present nuclear versus Navy 13-year fuel cycle.

Answer: Detailed information on the Navy fuel cycle is not available to Applicant. A trade-off and safety study cannot therefore be made.

Question No. D-94

Question: Furnish an assessment of the rupture of the reactor pressure vessel by brittle fracture.

Answer: Rupture of the reactor pressure vessel by brittle fracture is not possible because reactor vessel materials are not brittle during operation. See answer to Set A Question 17.

Question No. D-95

Question: Furnish a study of shock waves by failure of the reactor pressure vessel.

Answer: Failure of the reactor pressure vessel is not a postulated accident for this plant. Accordingly, no study of shock waves by such failure has been made for this plant. See response to Set D Question 94.

Question No. D-101

Question: Furnish an estimate of the fresh water needs of the complex.

Answer: ~17 million gallons per year of fresh water with Indian Point Units 1, 2 and 3 in operation.

Question No. D-104

Question: In view of the 1% limit imposed by Maryland on nuclear plants, would you agree to have your Technical Specifications and any operating conditions impose a limit on routine releases and off-site doses below 1%?

Answer: Applicant does not believe a % limit need be imposed on the Indian Point Unit No. 2 Technical Specifications. The proposed Technical Specifications have been written having in mind the "as low as practicable" philosophy. Several specifications have been written on operating procedures and availability of radwaste equipment which guarantee that normal operating releases will be as low as practicable (Normal operating releases are expected to be small fractions of those specified in 10CFR20).

At the same time, flexibility of operation, compatible with considerations of health and safety, is maintained to assure that the public is provided a dependable source of power even under unusual operating conditions.

Question No. D-105

Question: If not, what percent will you agree to stay below.

Answer: (See response to Question 104).

Question No. D-106

Question: The emergency electric power system consisting of three diesels does not conform to ACRS instructions and warnings about common mode failure. Why not employ a gas turbine and two diesels.

Answer: The three diesel generators represent only the on-site emergency power supply. Indian Point Unit No. 2 has, however, off-site power which is required by the Technical Specifications to be available and operable consisting of two 138 kv lines and a 13.8 kv line to Buchanan Substation.

Any of those sources, the on-site or any one off-site, are sufficient to supply emergency power to safeguards at Indian Point Unit No. 2. The diversity present in the Indian Point Unit No. 2 design gives assurance of protection fron consequences of common mode failures in the emergency power supply.

There is a 21mw gas turbine on site that can be readily used to supply power to Indian Point Unit No. 2 through a 13.8/6.9 kv transformer. The gas turbine, however, has a longer starting time than the diesels and, therefore, cannot be considered as a complete substitute emergency power source.

Question:No. D-107

Question: Are the diesels fed from a common tank?

Answer: No, three fuel oil storage tanks will be on site having a capacity of 7700 gallons each. Transfer of oil from an underground storage tank to automatically maintain level in each unit tank is accomplished by a motor-driven pump for each unit tank. Any oil transfer pump is capable of serving (see FSAR Section 8.2) any one or more emergency generator unit tanks through manual valving.

Question No. D-111

Question: Furnish progress reports of continuing research and development programs concerned with better determination of the efficiency of iodine removal systems. Parties to Indian Point Unit No. 3 construction license have not been furnished with new data (Page 26 Initial Decision, August 13, 1969), which would also be applicable to systems being employed in Indian Point Unit No. 2.

Answer: See the response to Set A Question 38. See also the progress report on research and development of iodine removal efficiency as provided in the Summary of Application, Section VII B.

Question No. D-112

Question: Furnish written progress reports of R & D programs concerned with the efficiency of impregnated charcoal filters for removal of organic iodides under postulated accident conditions.

Answer: See Page 6.4-5, 6.4-7, 6.4-17, 6.4-28; answer to Question 14.10 in the FSAR, and summary of Application Page 55.

Question No. D-114

Question: Describe progress of meteorological studies underway to determine conditions for radiation releases that are optimum.

Answer: The meteorological studies which are underway are to provide additional data covering the meteorological characteristics of the site. Wind observations made at a 100' meteorological tower at Indian Point and at a ship anchored in the Hudson River northwest of Indian Point in 1970 were compared with observations made at similar installations in 1955-56.

A summary of the findings follow:

1) Annual average statistics of wind speed, direction and vertical temperature difference were substantially the same for 1956 and 1970.

2) Average wind hodographs at the ships exhibited the same diurnal reversal pattern and the same 2.5m/sec night time downvalley speed in both years.

3) Persistance of low-speed unidirectional winds under inversion conditions was very weak.

4) Wind characteristics in 1970 were not significantly different than in 1956 (meteorological studies presented Section 2.6 of the Indian Point Unit No. 2 FSAR). Thus, the validity of the 1956 meteorological studies has been corroborated.

Question No. D-117

Question: Furnish progress reports of research on capability of impregnated charcoal filters to remove iodine at high humidity.

Answer: See response to Set D Question 112.

Question No. D-119

Question: Describe in detail the justification for believing combined spray, charcoal filter systems and plateout will enable Indian Point Unit No. 2 in post-accident to hold releases to 10CFR100 without taking credit for isolation water valve reduction of containment leakage.

Answer: Detailed justification is provided in Section 14.3.5 of the FSAR.

Question No. E-4

Question: What studies have been conducted to determine the feasibility of placing nuclear plants underground? Are these available? Have sites been investigated for possible use for underground plants?

Answer: We have made no studies of underground nuclear plants which determine their feasibility or safety advantages. Because of increased interest in such plants we have considered such a design for future plants and plan to investigate their feasibility and any benefits which may be associated with them.

Question No. E-6

Question: What factors about large reactors were considered in the AEC recommendation on May 23, 1959 (FED. Reg. - F. R. Doc. 56-4342) that large reactors be sited 10 to 20 miles from large cities? What specific advances have been made in design safety features since 1959 to now warrant disregard of these proposed guidelines?

Answer: (AEC response).

Question No. E-7

Question: Is the Indian Point Unit No. 2 plant an experiment or a proven workable plant design?

Answer: The question as stated calls for a conclusion which will vary depending on the definition of the terms used. However, in general, the design for Indian Point Unit No. 2 has been utilized in other plants which are operable, and hence in this sense is a "proven" design.

Question No. E-11

Question: The safety of nuclear power plants depends to a large extent upon the validity of the assumptions which underlie the AEC analysis. In a number of instances the AEC ass-umptions regarding the possibility of malfunction have been erroneous. Please identify those cases involving nuclear reactors where an event has occurred, with respect to the reactor was not predicted by the AEC review or produces more sever consequences than predicted. The incident at Enrico Fermi would be one example. The recent leaks at Indian Point Unit No. 1 would be a second.

Answer: (AEC response).

Question No. E-12

Question: With respect to the previous question please identify the actual cuase for the miscalculation and what if any steps have been taken to prevent such miscalculations.

Answer: (AEC response).

Question No. E-21

Question: What additional R & D and studies must Con Edison perform; what happens, if unsuccessful design modifications are required; will they increase rad discharge?

Answer: None, except that specified in response to Set B Question 20.

Question No. E-22

Question: Specify all cases where Con Edison says potential doses at site boundary would not exceed guidelines 10CFR100; how much actual dose will be.

Answer: Analyses presented in Section 14.0 of the Indian Point Unit No. 2 FSAR detail the doses resulting from all accidents considered. All doses given in Section 14 are calculated using extremely conservative assumptions. The actual doses would be far below those calculated.

Question No. E-23

Question: Are there any conditions under which off-site doses can exceed doses at the site boundary.

Answer: There are none for the Indian Point Unit No. 2 plant. For releases from IP-2 doses decrease with increasing distance.

Question No. E-24

Question: Specify all cases where potential doses would exceed 10CFR100.

Answer: There are no credible accidents for which potential doses would exceed 10CFR100. (see response to Question No. E-22.)

Question No. E-27

Question: Has the maximum credible accident included loss of source coolant water such as oil slick fire in Hudson River similar to Arthur Kill fire?

Answer: Applicant is not aware of any oil slick fire at Arthur Kill.

The maximum credible accident, the loss-of-coolant accident, does not consider the concurrent loss of Hudson River water as a coolant. As stated in response to Question 66, Set D, the loss of secondary coolant from the river has been analyzed in the FSAR, Section 14.1.9 and Appendix 14A, Section 5.2.

Question No. F-23

Question: During operation of the reactor, how will potential source of failure of the integrity of the primary coolant system be detected if they develop?

Answer: The proposed Technical Specifications, Section 4.2, Primary System Surveillance, has the objective of assuring the continued integrity of the primary system boundary. This specification applies to pre-operational and in-service structural surveillance of the reactor vessel and primary system boundary. Also, Section 3.1.F of the proposed Technical Specifications, Leakage of Reactor Coolant, describes the four reactor coolant leak detection systems based on three different principles, i.e., activity, humidity, and condensate flow measurements. Leakage through any of the walls of the primary system could be indicative of materials failure. Early detection of such leaks by the systems described in the Technical Specifications provides added assurance of maintaining the integrity of the primary system boundary.

Question No. F-25

Question: What precautions have been taken and will be taken in service to insure the integrity of the primary pump motor flywheel.

Answer: See response to Q. 4.2.1, Q. 4.2.2 and Q. 4.2.3 in the FSAR Vol. 5 as well as the Tech Specs, Section 4.2.

Question No. F-26

Question: Can the fuel element failure detection system be tested during reactor operation?

Answer: Yes, radiochemical analysis of the reactor coolant is required by the Technical Specifications and provides a check and calibration of the monitor.

Question No. F-28

Question: In an accident situation, what is the maximum calculated containment pressure and what are the assumptions made in calculating this pressure?

Answer: See answer to Set B Question 13.

Question No. F-29

Question: By what percentage does the containment design or pressure exceed the maximum calculated containment pressure?

Answer: 17.5%

Question No. F-34

Question: Has the Emergency Core Cooling System at Indian Point Unit No. 2 been tested or will it be tested before the reactor is operated?

Answer: The Emergency Core Cooling System at Indian Point Unit No. 2 will be tested before the reactor is operated.

Question No. F-38

Question: Why was a reactor pit crucible included in the initial design of Indian Point Unit No. 2 and why was it later decided to omit it?

Answer: See Section 1.1 of the FSAR, starting on p. 1.1 -3.

Question No. F-50

Question: When is it estimated that the post-accident hydrogen control system will be operational at Indian Point Unit No. 2.

Answer: The post-accident hydrogen control system for Indian Point Unit No. 2 consists of redundant Hydrogen flame recombiners as described in the FSAR in response to Q. 6. Vol. 5. These will be operational as required by the Technical Specifications, whenever the reactor is critical.

As an additional system for post-accident hydrogen control, Applicant has included provisions for controlled containment purge system which will be installed during the first two years of operation at power. (see testimony of John T. Stiefel)

Question No. F-59

Question: Is it possible that the use of a stack for the normal release of gaseous effluents from Indian Point Unit No. 2 would be superior to the plant vent presently planned from the standpoint of reduced ground level gaseous effluent concentrations in the vicinity of the plant site.

Answer: Use of the Indian Point Unit No. 1 stack to discharge Indian Point Unit No. 2 radioactive gas releases would result in doses lower than those resulting from an equivalent release through the Indian Point Unit No. 2 plant vent.

Question No. F-60

Question: In the event of an accident, would the use of a stack give any greater assurance that 10CFR100 would be complied with.

Answer: No. No means exist for collecting leakage from IP-2 and directing it to a stack.